NON-PUBLIC?: N
ACCESSION #: 9112300327
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Washington Nuclear Plant - Unit 2 PAGE: 1 OF 8

DOCKET NUMBER: 05000397

TITLE: REACTOR SCRAM AND ECCS INJECTION DUE TO FAILURE OF A FEEDWATER
LEVEL CONTROL SYSTEM COMPONENT
EVENT DATE: 11/19/91 LER #: 91-032-00 REPORT DATE: 12/19/91

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: 1 POWER LEVEL: 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: R. E. Fuller TELEPHONE: (509) 377-4148

COMPONENT FAILURE DESCRIPTION:
CAUSE: B SYSTEM: JB COMPONENT: CAP MANUFACTURER: B045
C AD FU
REPORTABLE NPRDS: Y
Y

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On November 19, 1991 at 0741, a reactor scram occurred as a result of Main Turbine Governor Valve Fast Closure initiated by a turbine trip signal from Reactor Pressure Vessel (RPV) high water level signal at Level 8 or +54.5 inches. Later, the High Pressure Core Spray (HPCS) System automatically injected as RPV level approached Level 2 (-50 inches). The high reactor water level was caused by a component failure in the non-safety related Feedwater Level Control (FWLC) System which resulted in the acceleration of the Reactor Feedwater (RFW) pumps in response to an erroneous feed-flow/steam-flow mismatch signal. The erroneous signal was caused by a failed capacitor in a Bailey Type 752 feed flow summation card in the FWLC System, which resulted in opening the 1.0 ampere circuit protection fuse. Low RPV level occurred later as a result of inadequate coolant makeup after the RFW pumps tripped at Level 8.

The immediate corrective action was prompt response by the Plant Operators to maneuver the Plant to a safe shutdown condition and to implement appropriate Plant procedures in a timely manner.

The root cause for failure of the Feed Water Level Control (FWLC) System was a failed capacitor in the feedwater summation circuitry.

Abstract (cont'd)

The corrective action was to replace the FWLC summer board RFW-SUM-615 with a tested board that has been checked for proper amperage draw.

The safety significance of this event is considered negligible because all safety systems were operable and responded as designed. Adequate core cooling was provided at all times. This event posed no threat to the safety of Plant personnel or the public.

END OF ABSTRACT

Plant Conditions

Power Level - 100%
Plant Mode - 1

Event Description

On November 19, 1991 at 0741, a reactor scram occurred as a result of Main Turbine Governor Valve Fast Closure initiated by a turbine trip signal from Reactor Pressure Vessel (RPV) high water level signal at Level 8 or +54.5 inches. Later, the High Pressure Core Spray (HPCS) System automatically injected as RPV level approached Level 2 (-50 inches). The high reactor water level was caused by a component failure in the non-safety related Feedwater Level Control (FWLC) System which resulted in the acceleration of the Reactor Feedwater (RFW) pumps in response to an erroneous feed-flow/steam-flow mismatch signal. The erroneous signal was caused by a failed capacitor in a Bailey Type 752 feed flow summation card in the FWLC System, which resulted in opening the 1.0 ampere circuit protection fuse. Low RPV level occurred later as a result of inadequate coolant makeup after the RFW pumps tripped at Level 8.

The effect of fuse failure was to cause the summed feedwater flow signal input to the FWLC circuit to indicate zero feedwater flow. This resulted in a maximum feedwater flow demand signal to the RFW pumps and a transfer of the Reactor Recirculation (RRC) pumps from 60 Hz to 15 Hz (fast to slow speed) after a 10 second time delay per design. The decrease in RRC flow resulted in an increase in coolant void formation and a subsequent increase in RPV level. The increased feed to the Reactor Pressure Vessel (RPV) combined with the swell in RPV level from decreased RRC flow caused a level transient which tripped the Main Turbine and the RFW pumps on a Level 8 (+54.5 inches) signal at 07:41:55. The Main Turbine trip at greater than 304; power initiated a Reactor Protection (RPS) System scram due to a decrease in oil supply pressure to the turbine governor valves to less than 1250 psig.

Reactor power at the time of the scram had decreased to approximately 40% as a result of transfer of the RRC pumps to slow speed. RPV level peaked at +58 inches (Narrow Range) and began decreasing rapidly because the Main Turbine Bypass System was drawing off reactor inventory in the form of steam to control reactor pressure. At approximately -20 inches the Reactor Core Isolation Cooling (RCIC) System was manually initiated to regain level control.

Level continued to drop and at 07:42:31 the Main Steam Isolation Valves (MSIVs) closed and other isolations occurred, including closure of Reactor Closed Cooling (RCC) water isolation valves to Drywell cooling when the RPV level reached -43 inches on Wide Range (WR) instrumentation. At 07:42:43, HPCS initiated with RPV level at -46 inches on WR instrumentation. The lowest RPV level was -48 inches at 07:42:48. Level was then restored by RCIC and HPCS. RRC pump 1A received a Level 2 ATWS trip signal, but RRC pump 1B did not trip because its logic was not satisfied. (Level did not drop below the minimum allowable trip level of -50 inches.)

An Unusual Event was declared at 07:47 due to HPCS Level 2 initiation even though the minimum RPV level indication was -48 inches instead of -50 inches required by the notification procedure. The appropriate outside agency notifications and Plant announcements were made, including the required immediate notifications via the Emergency Notification System (ENS). HPCS and RCIC restored RPV level to +54 inches. HPCS was then secured and placed in standby status. RCIC injection was inhibited. With the subsequent expansion of the cooler HPCS water, the RPV level peaked at 77 inches. Direction was then given by the Control Room Supervisor to maintain pressure between 800 and 1000 psig with the Safety Relief Valves (SRV) and RPV level between +13 inches and +54.5 inches with the RCIC system. Also, the RPS System scram was reset and cooling to the control rods was re-established.

Drywell cooling had stopped with closure of RCC Drywell cooling isolation valves. Heat was still being transferred to the Drywell from the RPV and associated piping and components. This caused an increase in Drywell pressure and temperature. At 08:01:44 a high drywell pressure signal was received and HPCS re-initiated. Injection was manually prevented by securing the HPCS pump because level was being maintained with RCIC.

While attempting to maintain level, SRV cycling caused the RPV level to fall to +11 inches and resulted in a second RPS scram signal at 08:15. The Control Room Operators (CRO) responded appropriately to the second Reactor Scram and reset the RPS. Recovery from the Drywell high pressure trip was then initiated to re-establish drywell cooling.

The RFW System was reconfigured to route feedwater through the Startup Flow Control Valves (RFW-FCV-10A and RFW-FCV-10B), and the Digital Electro-Hydraulic Control (DEH) System was programmed to reduce RPV pressure within the cool down limits. At 08:52 the CROs equalized pressure across the MSIVs. They were then opened to re-establish the condenser as a heat sink. The DEH controlled Main Turbine Bypass valves were used to depressurize the RPV to about 500 psig to facilitate using the condensate booster pumps for level control. The plant secured from the Unusual Event at 09:45.

Immediate Corrective Action

The Plant Operators responded promptly to maneuver the Plant to a safe shutdown condition. Appropriate Plant procedures were implemented in a timely manner during response to the transient conditions of the Plant.

Further Evaluation and Corrective Action

A. Further Evaluation

1. This event is reportable per 10CFR50.73 (a)(2)(iv) as a condition that resulted in automatic actuation of the Reactor Protection System (RPS) and an Engineered Safety Feature (ESF). Failure of a capacitor in a FWLC System summer circuitry caused the circuit protective fuse to open. This resulted in a high flow demand signal to the RFW pumps which caused a Main Turbine trip on RPV high level and a resulting RPS actuation. The high level caused the RFW turbines to trip which resulted in low RPV

level and subsequent HPCS initiation. Also, a high Drywell pressure resulted in a second HPCS initiation and other selected ESF actuations.

2. There were no other structures, components, or systems inoperable prior to the event which contributed to the event.

3. The root cause for failure of the FWLC System which led to the reactor scram was a component failure in the FWLC circuitry. Investigation determined capacitor C-3 on FWLC summer card RFW-SUM-615 failed in the shorted condition. Excessive amperage draw from the failed capacitor caused the 1 amp circuit protective fuse to open, resulting in a zero feed flow signal to the FWLC System. This caused the erroneous steam-flow/feed-flow mismatch and resultant high flow demand signal to the RFW pumps.

The capacitor which shorted and caused the fuse to blow on the RFW-SUM-615 board was a tantalum capacitor which experience has shown "shorts" when it fails. These type of capacitors are reliable capacitors with a mean life of 348 years and this particular capacitor was manufactured on the 25th week of 1973. From this information, the failure of this capacitor is considered to be premature.

4. Failure of the RFW-SUM-615 output to zero satisfied the RRC pump logic which requires transfer from fast to slow speed after a 10 second time delay if the feed flow is less than 3096. While this transfer was in progress, a reactor scram occurred from the Main Turbine trip on high RPV level, Level 8. Both RRC pumps were at slow speed when a Level 2 ATWS signal caused RRC-P-1A to trip off as designed. Level was within the trip setpoint tolerance band. However, RRC-P-1B remained at slow speed because RPV level did not go low enough to reach the trip setpoint for this pump, but the setpoint was also within the tolerance band. The RPV water level reached -48 inches versus the Level 2 setpoint of -50 inches. Instrument calibration for both trains of RRC were verified to be within the acceptable limits.

5. Following the transient, the RRC flow control valve RRC-FCV-60A indicated partially closed because of a blown fuse on the

power to the control system. The FCV valve actual position for RRC-P-1A remained unchanged in the open position. When power to the control system is lost, the flow control valve locks in position without any movement. Failure of the fuse occurred coincident with the transient, but was completely unrelated to the initiating event and provided no influence on the progression or outcome of the transient.

6. Using instrument tolerances provided by the Master Data Sheets and the design logic diagrams, it was determined that all NSSSS, ECCS, RRC, RCIC and RFW System responses were consistent with design expectations. Reactor conditions for RPV Level 2 and high Drywell pressure entered the instrument trip setpoint tolerance band but did not exceed the allowable setpoint. Consequently, the trip logic for all components within the above described systems was not satisfied, and as a result, only part of the components received trip/actuation signals. However, all components receiving trip/actuation signals performed as designed. And those that did not trip were not required to trip nor should have tripped to mitigate the consequences of the accident. Therefore, all systems performed as designed with consideration of the component failures identified.

7. In accordance with the requirements of Technical Specifications 3.5.1, Action (f) and 6.9.2 , the following special data is provided:

- Total accumulated ECCS initiation cycles to date equals five (5).

- Current usage factor value remains below 0.70.

B. Further Corrective Action

1. The FWLC summer board RFW-SUM-615 was replaced with a tested board that has been checked for proper amperage draw.

Safety Significance

The safety significance of this event is considered negligible because all safety systems were operable and responded as designed. This event was well within the FSAR transient analyses. Adequate core cooling was provided at all times. This event posed no threat to the safety of Plant personnel or the public.

Similar Events

LER 87-02 documented a similar event that occurred on March 22, 1987 in which the circuit protection fuse failed on the feed flow summer board RFW-SUM-615 of the FWLC System. This caused a zero feed flow signal to the FWLC System and corresponding high flow demand signal to the RFW pumps. The cause of the fuse failure was investigated but was not determinable. Corrective actions to this problem was replacement of the 1/4 amp fuse on this board and boards RFW-SUM-602, 603, 615, and 616 with a 1 amp fuse to increase reliability. Then on October 1, 1991 this same fuse was found blown during start up operations and the fuse was replaced with a 1 amp fuse. Failure of the fuse at this time was attributed to the de-term and re-term work performed in this cabinet during the R-6 outage.

EIIS Information

Text Reference EIIS Reference
System Component

Reactor Containment C
Control Rod Drive System AA
Turbine Supervisory Control System JJ
Main Turbine Control Fluid System TG
Emergency Power System for HPCS DG
High Pressure Core Spray System BG
Reactor Recirculation System AD FU
Low Pressure Core Spray System BM
Incore/Excore Monitoring System IG
Main Steam System SB
Turbine Bypass System SO
Reactor Building NG
Reactor Protection System JC
Reactor Building Closed Cooling Water System CC
Reactor Core Isolation Cooling System BN
Feedwater Pump Turbine I&C System JK
Reactor Feedwater Control System JB CAP
Feedwater System SJ
Feedwater Pump Injection System SK
RHR/Containment Spray BO
Containment & Reactor Vessel Isolation System JM
Main Turbine System TG

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WASHINGTON PUBLIC POWER SUPPLY SYSTEM

P.O. Box 968 o 3000 George Washington Way o Richland, Washington 99352

Docket No. 50-397
December 19, 1991
G02-91-231

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Subject: NUCLEAR PLANT NO. 2
LICENSEE EVENT REPORT NO. 91-032

Dear Sir:

Transmitted herewith is Licensee Event Report No. 91-032 for the WNP-2 Plant. This report is submitted in response to the report requirements of 10CFR50.73 and discusses the items of reportability, corrective action taken, and action taken to preclude recurrence.

Very truly yours,

J. W. Baker (M/D 927M)
WNP-2 Plant Manager

Enclosure:
Licensee Event Report No. 91-032

cc: Mr. John B. Martin, NRC - Region V
Mr. C. Sorensen, NRC Resident Inspector (M/D 901A)
INPO Records Center - Atlanta, GA
Ms. Dottie Sherman, ANI
Mr. D. L. Williams, BPA (M/D 399)
NRC Resident Inspector - walk over copy

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